AmerGen
An Exelon Company

Clinton Power Station
R. R. 3, Box 228
Clinton, IL 61727

10 CFR 50.73

U-603685

August 16, 2004

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Licensee Event Report 2004-001-01

Enclosed is Licensee Event Report (LER) No. 2004-001-01: Generator Neutral Over-Voltage / Lockout Leads to Reactor Scram. This supplemental report is being submitted in accordance with the requirements of 10CFR50.73.

Should you have any questions concerning this report, please contact Mr. William Iliff, Regulatory Assurance Manager, at (217)-937-2800.

Respectfully,

R. S. Bement
Site Vice President
Clinton Power Station

JLP/blf

Enclosure: Licensee Event Report 2004-001-01

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety – IEMA Division of Nuclear Safety

IE22

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Clinton Power Station	05000461	1 OF 4

4. TITLE

Generator Over-Voltage / Lockout Leads to Reactor Scram

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									None	05000
03	22	2004	2004	- 001 -	01	08	16	04	FACILITY NAME: None	DOCKET NUMBER: 05000

9. OPERATING MODE	1
10. POWER LEVEL	093

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

20.2201(b)	20.2203(a)(3)(ii)	50.73(a)(2)(ii)(B)	50.73(a)(2)(ix)(A)
20.2201(d)	20.2203(a)(4)	50.73(a)(2)(iii)	50.73(a)(2)(x)
20.2203(a)(1)	50.36(c)(1)(i)(A)	X 50.73(a)(2)(iv)(A)	73.71(a)(4)
20.2203(a)(2)(i)	50.36(c)(1)(ii)(A)	50.73(a)(2)(v)(A)	73.71(a)(5)
20.2203(a)(2)(ii)	50.36(c)(2)	50.73(a)(2)(v)(B)	OTHER
20.2203(a)(2)(iii)	50.46(a)(3)(ii)	50.73(a)(2)(v)(C)	Specify in Abstract below or in NRC Form 366A
20.2203(a)(2)(iv)	50.73(a)(2)(i)(A)	50.73(a)(2)(v)(D)	
20.2203(a)(2)(v)	50.73(a)(2)(i)(B)	50.73(a)(2)(vii)	
20.2203(a)(2)(vi)	50.73(a)(2)(i)(C)	50.73(a)(2)(viii)(A)	
20.2203(a)(3)(i)	50.73(a)(2)(ii)(A)	50.73(a)(2)(viii)(B)	

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
R. W. Chickering, Engineering	(217) 937-2818

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	EL	CBL	G080	Y	B	EL	EXJ	P295	Y

14. SUPPLEMENTAL REPORT EXPECTED			15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On March 22, 2004, at about 1931 hours, an automatic reactor scram occurred with the plant at 93 percent power. Operators received a trouble alarm in the Main Control Room for the Hydrogen and Stator Cooling Cabinet followed by a Main Generator neutral over-voltage trip and Generator Trip System 2 Lockout. The Generator trip caused a Main Turbine trip and Turbine Control Valve fast closure, resulting in an automatic reactor scram. All control rods fully inserted. Following the reactor scram, reactor water level dropped, as expected, to the Low Level 3 trip setpoint, initiating the Reactor Protection System. Operators entered Emergency Operating Procedures due to the low reactor water level transient. The root cause investigation determined that a failed Partial Discharge Analysis (PDA) system cable and/or a piece of aluminum laminate from a degraded bus conductor expansion joint contacted the grounded bus duct causing a ground fault and subsequent generator neutral over-voltage trip. The failures of the PDA cable and the aluminum laminate were both due to mechanical fatigue caused by an increase in design air flow rate in the bus duct implemented during the Spring 2004 refueling outage. Corrective actions include implementing a design change for the bus duct expansion joint that is not susceptible to fatigue failure.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

A. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

Unit: 1 Event Date: 3/22/2004 Event Time: 1931 Central Standard Time

Mode: 1 (POWER OPERATION) Reactor Power: 93 percent

B. DESCRIPTION OF THE EVENT

On March 22, 2004, at about 1931 hours, an automatic reactor scram occurred with the plant at 93 percent power. Operators in the Main Control Room received a trouble alarm for the Hydrogen and Stator Cooling Cabinet [TK] [TJ] [CAB] followed by a Main Generator [GEN] [TB] neutral over-voltage trip and Generator Trip System 2 Lockout. The Generator trip caused a Main Turbine [TRB] [TA] trip and Turbine Control Valve [V] fast closure, resulting in an automatic reactor scram. All control rods fully inserted.

Following the scram, as expected, reactor pressure vessel water level dropped below the Low Level 3 trip setpoint to 0.0 inches Narrow Range, initiating the Reactor Protection System [JC]. (Low Level 3 is 8.9 inches Narrow Range indication.) At 1932 hours operators entered the actions of procedure CPS 4001.01, "Reactor Scram Off-Normal," in response to the reactor scram and the lowering RPV water level. At 1939 hours, operators completed the immediate actions of the reactor scram off-normal procedure and entered Emergency Operating Procedure (EOP) 1, "RPV Control," due to the low reactor water level transient.

At 2006 hours, the reactor scram signal was reset.

At 2215 hours, operators exited EOP 1 and transitioned into Procedure CPS 3006.01, "Unit Shutdown," as reactor pressure was stable at 858 psig and reactor water level was stable at 34 inches Narrow Range. At 0130 hours on March 23, operators exited the reactor scram off-normal procedure.

As expected, the Low Level 3 RPV water level trip caused Primary Containment Isolation Valves [ISV] in Group 2 (Residual Heat Removal (RHR) [BO]), Group 3 (RHR), and Group 20 (miscellaneous systems) to receive signals to shut; these valves were already shut prior to the event in accordance with the normal plant lineup.

The reactor remained in Mode 3 (HOT SHUTDOWN) with reactor coolant pressure being controlled between 800 and 1065 psig using the Turbine Bypass Valves and steam drains [DRN], and reactor coolant level being maintained between Low Level 3 and High Level 8 using the Motor-Driven Reactor Feed Pump [MO] [P] [SJ].

No Main Steam Isolation Valves closed and no Safety Relief Valves lifted during this event.

Condition Report 210033 was initiated to track the investigation and resolution of this event.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

B. DESCRIPTION OF THE EVENT (continued):

No automatic or manually initiated safety system actuations were necessary to place the plant in a safe and stable condition. No other inoperable equipment or components directly affected this event.

C. CAUSE OF THE EVENT

A complex troubleshooting plan was initiated to investigate this event. Troubleshooting identified two components within the Isolated Phase Bus Duct Cooling System [BDUC] [EL] failed. A Partial Discharge Analysis (PDA) System cable [CBL] was severed and a conductor expansion joint [EXJ] was torn during the initial inspection of the 'B' Isolated Phase Bus Duct.

The root cause investigation determined that the failed PDA cable and/or a piece of aluminum laminate from a degraded bus conductor expansion joint contacted the grounded bus duct and caused a ground fault and subsequent generator neutral over-voltage trip. The failures of the PDA cable and the aluminum laminate were both due to mechanical fatigue (vibration fatigue) caused by an increase in design air flow rate (from 21,700 scfm to 36,700 scfm) in the bus duct implemented during the Spring 2004 refueling outage. The increased air flow was implemented in order to accommodate increased cooling requirements for Extended Power Uprate.

D. SAFETY ANALYSIS

There were no actual safety consequences associated with his event. The event was reviewed for analyzed transients discussed in Chapter 15 of the Clinton Power Station Updated Safety Analysis Report. The analysis determined that this event was within the design basis of the plant.

No safety system functional failures occurred during this event.

E. CORRECTIVE ACTIONS

All three Isolated Phase Bus Ducts were inspected for degradation, the PDA System cables have been removed from all three Isolated Phase Bus Ducts, and an expansion link on the conductor in the 'B' Isolated Phase Bus Duct has been repaired. A design change will be implemented for the bus duct expansion joint that is not susceptible to fatigue failure.

F. PREVIOUS OCCURRENCES

Based on a review of industry operating experience, no previous similar events are known.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model	Mfg. Part Number
H.K. Porter	22 kV bus duct	n/a	n/a
General Electric	Partial Discharge Assembly Kit Coaxial Connector	357A2528P0003	n/a

Note: The H.K. Porter isolated phase bus duct product line was purchased by Delta-Unibus.